Progress Energy

JUN 0 8 2005

SERIAL: BSEP 05-0072

10 CFR 50.73

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Brunswick Steam Electric Plant, Unit No. 2
Docket No. 50-324/License No. DPR-62
Licensee Event Report 2-2005-002

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc., submits the enclosed Licensee Event Report. This report fulfills the requirement of a written report with sixty days of a reportable occurrence.

Please refer any questions regarding this submittal to Mr. Edward T. O'Neil, Manager – Support Services, at (910) 457-3512.

Sincerely,

David H. Hinds
Plant General Manager
Brunswick Steam Electric Plant

GLM/glm

Enclosure:

Licensee Event Report

Progress Energy Carolinas, Inc.
Brunswick Nuclear Plant
P.O. Box 10429
Southport, NC 28461

IE22

cc:

U. S. Nuclear Regulatory Commission, Region II
ATTN: Dr. William D. Travers, Regional Administrator
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, GA 30303-8931

U. S. Nuclear Regulatory Commission
ATTN: Mr. Eugene M. DiPaolo, NRC Senior Resident Inspector
8470 River Road
Southport, NC 28461-8869

U. S. Nuclear Regulatory Commission **(Electronic Copy Only)**
ATTN: Ms. Brenda L. Mozafari (Mail Stop OWFN 8G9)
11555 Rockville Pike
Rockville, MD 20852-2738

Ms. Jo A. Sanford
Chair - North Carolina Utilities Commission
P.O. Box 29510
Raleigh, NC 27626-0510

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Brunswick Steam Electric Plant (BSEP), Unit 2	05000324	1 OF 5

4. TITLE

Automatic Shutdown Due to Condensate System Transient

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
04	09	2005	2005 --	002 --	00	06	08	2005	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more)			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	X 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
065	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	OTHER Specify in Abstract below or in NRC Form 366A
	☐ 20.2203(a)(2)(vi)	X 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Gary Miller, Lead Engineer - Licensing	(910) 457-2110

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MO	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On April 9, 2005, Unit 2 was operating at 65 percent of rated thermal power. At 0049 hours, a Condensate system transient tripped the Condensate Booster Pumps and the running 2B Reactor Feed Pump (RFP). Vessel water level lowered, resulting in an automatic reactor protection system actuation at 0050 hours. Vessel water level was restored with the 2A RFP, the Reactor Core Isolation Cooling system, and Control Rod Drive system flow. All appropriate system isolations and actuations occurred as designed. After the shutdown, Reactor Coolant system (RCS) heatup rates were exceeded; however this was not recognized at the time of occurrence and an engineering evaluation was not performed, per Technical Specification (TS) Limiting Condition for Operation (LCO) 3.4.9 Required Action A.2, until after Unit 2 had changed operating modes to commence restart and power operation. This is prohibited by TS LCO 3.0.4.

The cause of the shutdown was a failure to establish Condensate system flow limitations, which resulted in failure to maintain adequate pressure to the suction of the RFPs. The cause of the failure to evaluate RCS heatup rates prior to mode change was attributed to procedures that were deficient in addressing the operating configuration that occurred after the shutdown. Applicable operating procedures will be revised to provide appropriate guidance. The safety significance of this event is minimal because vessel water level was restored, all appropriate safety system actuations occurred, and the RCS was evaluated as acceptable for continued operation.

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant (BSEP), Unit 2	05000324	2005	-- 002 --	00	2 OF 5

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Energy Industry Identification System (EIIS) codes are denoted in the text as [XX].

INTRODUCTION

On April 9, 2005, Unit 2 was in power ascension, operating at 65 percent of rated thermal power (RTP). At 0049 hours, a Condensate system [SD] transient resulted in tripping the Condensate Booster Pumps (CBP) and the running 2B Reactor Feed Pump (RFP)/[SJ]. Vessel level lowered to the Reactor Protection system (RPS)/[JC] trip setpoint, resulting in an automatic reactor shutdown at 0050 hours. Vessel water level was restored with the 2A RFP, the Reactor Core Isolation Cooling system (RCIC)/[BN] and flow from the Control Rod Drive system (CRD)/[AA]. Prior to the event, there were no safety systems that were inoperable or that contributed to the event. During the event, all appropriate system isolations and actuations occurred as designed. At 0453 hours, notification was made to the NRC (i.e., Event Number 41582) in accordance with 10 CFR 50.72(b)(2)(iv)(B), and 10 CFR 50.72(b)(3)(iv)(A). This event is being reported in accordance with 10 CFR 50.73(a)(2)(i)(B), as operation prohibited by Technical Specifications (TS), and 10 CFR 50.73(a)(2)(iv)(A), automatic actuation of specified systems, as described below.

EVENT DESCRIPTION

On April 9, 2005, at 0049 hours, with Unit 2 operating at 65 percent of RTP, the 2A RFP speed was reduced to support power ascension testing, and the pump was no longer feeding the reactor vessel. Immediately after 2A RFP flow was terminated, an unanticipated low suction pressure in the Condensate piping occurred that tripped the CBPs and the 2B RFP. After the 2B RFP tripped, actions were taken to return the 2A RFP to service, but the suction pressure to the pump had not yet been completely restored when vessel water level lowered to the Low Level 1 RPS trip setpoint, resulting in an automatic reactor shutdown at 0050 hours. All control rods inserted, both Reactor Recirculation (RR)/[AD] pumps tripped, the Alternate Rod Injection system [JC] actuated, the Reactor Building Ventilation system [VA] isolated, and the Standby Gas Treatment system [BH] started. In addition, the following Primary Containment Isolation system (PCIS)/[JM] actuations occurred: Group 2 (i.e., Drywell Equipment and Floor Drain [WK], Traversing In-core Probe [IG], Residual Heat Removal (RHR)/[BO] Discharge to Radwaste, and RHR Process Sample [KN] isolation valves), Group 3 (i.e., Reactor Water Cleanup (RWCU)/[CE] system isolation valves), and Group 6 (i.e., Containment Atmosphere Control/Dilution [BB], Containment Atmosphere Monitoring [IK], and Post Accident Sampling [IP] system isolation valves). All appropriate system isolations and actuations occurred as designed. When vessel water level dropped to the Low Level 2 setpoint, the High Pressure Coolant Injection (HPCI)/[BJ] system and RCIC received automatic initiation signals. Vessel water level was restored with the 2A RFP, RCIC, and CRD flow. HPCI did not inject because reactor vessel water level was restored above the Low Level 2 setpoint prior to the HPCI injection valve opening.

The investigation of the shutdown identified a timing problem with the Digital Feedwater Control system which is being reported in LER 1-2005-001.

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant (BSEP), Unit 2	05000324	2005	-- 002 --	00	3 OF 5

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

EVENT DESCRIPTION (continued)

With the unit in Mode 3 (i.e., Hot Shutdown) following the shutdown, the cooldown rate at the reactor vessel bottom head and RR loop A exceeded the limit of 100 degrees F in any one hour period as required by TS Surveillance Requirement 3.4.9.1.b. TS 3.4.9 Limiting Condition for Operation (LCO) Required Action A.2, requires a determination, to be performed within 72 hours, that the Reactor Coolant system (RCS) is acceptable for operation. An engineering evaluation was performed within the required timeframe, and it concluded that the temperature changes in the lower head region of the reactor vessel and RR loop A did not violate any safety margins of the American Society of Mechanical Engineers (ASME) Code, Section XI or 10 CFR 50, Appendix G (i.e., the RCS was acceptable for continued operation.) In two other instances, RCS heatup rates were exceeded but the conditions were not recognized until April 14, 2005, and therefore, an engineering evaluation was not performed in the required timeframe. The first instance occurred on April 9, 2005 at approximately 0520 hours while the operating crew was warming the RHR discharge piping in preparation for the shutdown cooling mode. Based on review of operating data, the heatup rate was found to be 139 degrees F/hr on RR loop A and 146 degrees F/hr on RR loop B. The second instance occurred on April 9, 2005, at approximately 1020 hours while the operating crew was raising and lowering RHR flow to establish shutdown cooling. A thermal stratification had formed in the lower vessel region after the shutdown. When the thermal layer dispersed, the bottom head drain temperature was found to have increased at a rate of approximately 131 degrees F/hr.

Unit 2 entered Mode 4 (i.e., Cold Shutdown) on April 9, 2005, at 1635 hours and entered Mode 2 (i.e., Startup) on April 11, 2005, at 1207 hours during the restart sequence. Neither heatup rate exceedence was recognized until operating data was reviewed in the corrective action program on April 14, 2005. Because Unit 2 changed operating modes on April 11, 2005, with this condition unrecognized, an engineering evaluation was not performed in accordance with TS 3.4.9 Required Action A.2, until after Unit 2 had changed operating modes to commence restart and power operation. This is prohibited by TS LCO 3.0.4. which states, "When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made ..." Subsequent engineering evaluations determined that temperature changes in the lower head region of the reactor vessel and RR loops did not violate any safety margins of the ASME Code, Section XI or 10 CFR 50, Appendix G.

EVENT CAUSE

The root cause of the shutdown was that the operating strategy of the Condensate system was to use a pump discharge pressure band for control without establishing a flow limit. The Condensate and Feedwater systems had undergone modifications during the spring 2005 refueling outage that changed flow and pressure operating characteristics. The modifications included a detailed hydraulic analysis and a methodical startup testing process, including inspections, hold points, data analysis, and evaluation of anomalies. At the time the automatic RPS actuation occurred, the Condensate system discharge pressure was being controlled within a pressure band of 150 to 190 psig, as required by a plant operating procedure. This was being accomplished by diverting flow to the condenser via operation of a Condensate return valve. Also, with the 2A RFP removed from service, the associated minimum flow valve was opened, which diverted additional flow to the condenser.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant (BSEP), Unit 2	05000324	2005	-- 002 --	00	4 OF 5

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

EVENT CAUSE (continued)

No procedural guidance existed for limiting the Condensate system flow rate. As a result, overall Condensate system flow exceeded the capability of the Condensate system. This condition created a low suction pressure in the Condensate piping which led to the tripping of the CBPs and the 2B RFP.

The root cause of the failure to evaluate RCS heatup and cooldown rates prior to mode change was attributed to deficient procedures for monitoring vessel heatup or cooldown activities. The post-shutdown conditions caused thermal stratification in the lower vessel region. Recovery from this condition led to heatup rates in the RR piping and the vessel bottom head drain line that exceeded the TS limit. The vessel bottom head metal temperature was monitored, and it did not exceed the TS limit; however, the vessel drain temperature indicated that the TS temperature heatup rate limit was exceeded. In this mode of operation, this temperature point and the RR loop temperature point were not monitored for heatup rate based on plant procedural guidance that was developed based on industry guidelines. Consequently, these two instances were not immediately identified, and not evaluated in a timely manner.

CORRECTIVE ACTIONS

RPS and Subsequent System Actuations:

Applicable operating procedures will be revised to include Condensate system flow guidance to prevent low suction pressure conditions in the condensate piping.

Failure to Evaluate RCS Heatup Rates Prior to Mode Change:

Applicable operating procedures will be revised to include appropriate guidance on temperature monitoring.

SAFETY ASSESSMENT

The safety significance of the event is minimal because vessel water level was promptly restored and all appropriate safety system actuations occurred. The safety significance of the RCS heatup rate exceedence omission was minimal based on the conclusion of the subsequent engineering evaluation.

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant (BSEP), Unit 2	05000324	2005	-- 002	-- 00	5 OF 5

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

PREVIOUS SIMILAR EVENTS

LER 1-2003-001: Reactor Feed Pump Trip Results in Specified Systems Actuation
On January 12, 2003, a reactor feedwater pump turbine (RFPT) trip, resulted in the actuation of the RPS, as well as PCIS Group 2 and 6 valve closures. At the time of the event, Unit 1 was initially operating at 94 percent of RTP. The cause of the RPS and subsequent equipment actuations was attributed to insufficient lube oil pressure on the RFPT 1B bearing header during an oil pump trip.

The corrective actions in this LER involve equipment failures, and therefore, could not be expected to prevent the event described in LER 2-2005-002.

COMMITMENTS

There are no commitments in this LER. The actions discussed within this report will be implemented in accordance with corrective action program requirements.